UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
License No. DPR-40

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Omaha Public Power District (the licensee) notarized March 21 and August 3, 1978, as revised by letters dated October 31, November 7 and 27, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 43 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 5, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Replace the following pages and figures of the Appendix "A" Technical Specifications with the enclosed pages and figures. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
ii	ii
iii	iii
2	2
1-2	1-2
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DEFINITIONS

REACTOR OPERATING CONDITIONS (Continued)

Cold Shutdown Condition (Operating Mode 4)

The reactor coolant T_{cold} is less than 210°F and the reactor coolant is at shutdown boron concentration.

Refueling Shutdown Condition (Operating Mode 5)

The reactor coolant is at refueling boron concentration and T_{cold} is less than 210°F.

Refueling Operation

Any operation involving the shuffling, removal, or replacement of nuclear fuel, CEA's, or startup sources.

The Refueling Boron Concentration

A reactor coolant boron concentration of at least 1700 ppm, which corresponds to a shutdown margin of not less than 5% with all CEA's withdrawn.

Shutdown Boron Concentration

The boron concentration required to make the reactor subcritical by the amount defined in paragraph 2.10.

Refueling Outage or Refueling Shutdown

A plant outage or shutdown to perform refueling operations upon reaching the planned fuel depletion for a specific core.

Plant Operating Cycle

The time period from a Refueling Shutdown to the next Refueling Shutdown.

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.1 Safety Limits - Reactor Core (Continued)

(DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNBR of 1.30 corresponds to a 95% probability at a 95% confidence level that DNB will not occur, which is considered an appropriate margin to DNB for all operating conditions.(2)

The curves of Figures 1-1, 1-2, and 1-3 represent the loci of points of reactor thermal power (either neutron flux instruments or ΔT instruments), reactor coolant system pressure and cold leg temperature of various pump combination for which the DNBR is 1.30. The area of safe operation is below these lines. For 3 and 2 pump operation, the limiting condition is void rather than the DNBR. The void fraction limits assure stable flow and maintenance of DNBR greater than 1.3.

The reactor core safety limits are based on radial peaks limited by the CEA insertion limits in Section 2-10 and axial shapes within the axial power distribution trip limits in Figure 1-4 and a total unrodded planar radial peak of 1.62. The LSSS in Figures 1-5, 1-6, and 1-7 are based on the assumption that the unrodded integrated total radial peak (F_r^T) is 1.57. This peaking factor is slightly higher (more conservative) than the maximum predicted unrodded total radial peak during core life, excluding measurement uncertainty.

Flow maldistribution effects for operation under less than full reactor coolant flow have been evaluated via model tests.(3) The flow model data established the maldistribution factors and hot channel inlet temperature for the thermal analyses that were used to establish the safe operating envelopes presented in Figures 1-1, 1-2, and 1-3. These figures were established on the basis that the thermal margin for part-loop operation should be equal to or greater than the thermal margin for normal operation. The reactor protective system is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure and thermal power level that would result in a DNBR of less than 1.30.(4)

References

(1) FSAR, Section 3.5.5
(2) FSAR, Section 3.5.2
(3) FSAR, Section 1.4.6
(4) FSAR, Section 3.5.3

a. One of the operable charging pumps may be removed from service provided two charging pumps are operable within 24 hours.

b. Both boric acid pumps may be out of service for 24 hours.

c. One concentrated boric acid tank may be out of service provided a minimum of 68 inches of 6-1/4 percent to 12 percent by weight boric acid solution at a temperature of at least 20°F above saturation temperature is contained in the operable tank and provided that the tank is restored to operable status within 24 hours.

d. Only one flow path from the concentrated boric acid tanks to the reactor coolant system may be operable provided that either the other flow path from the concentrated boric acid tanks to the reactor coolant system or the flow path from the SIRW tank to the charging pumps is restored to operable status within 24 hours.

e. One channel of heat tracing may be out of service provided it is restored to operable status within 24 hours.

f. One level instrument on each concentrated boric acid tank may be out of service for 24 hours.

Basis

The chemical and volume control system provides control of the reactor coolant system boron inventory.[1] This is normally accomplished by using any one of the three charging pumps in series with one of the two boric acid pumps. An alternate method of boration will be to use the charging pumps directly from the SIRW storage tank. A third method will be to depressurize and use the safety injection pumps. There are two sources of borated water available for injection through three different paths.

(1) The boric acid pumps can deliver the concentrated boric acid tank contents (6-1/4 - 12 weight percent concentration of boric acid) to the charging pumps. The tanks are located above the charging pumps so that the boric acid will flow by gravity without being pumped.

(2) The safety injection pumps can take suction from the SIRW tank (at least 1700 ppm boron solution).

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System

Applicability

Applies to the operating status of the emergency core cooling system.

Objective

To assure operability of equipment required to remove decay heat from the core.

Specifications

(1) Minimum Requirements

The reactor shall not be made critical unless all of the following conditions are met:

a. The SIRW tank contains not less than 283,000 gallons of water with a boron concentration of at least 1700 ppm at a temperature not less than 40°F.

b. One means of temperature indication (local) of the SIRW tank is operable.

c. All four safety injection tanks are operable and pressurized to at least 240 psig with a tank liquid of at least 116.2 inches (67%) and a maximum level of 128.1 inches (74%) with refueling boron concentration.

d. One level and one pressure instrument is operable on each safety injection tank.

e. One low-pressure safety injection pump is operable on each bus.

f. One high-pressure safety injection pump is operable on each bus.

g. Both shutdown heat exchangers and three of four component cooling heat exchangers are operable.

h. Piping and valves shall be operable to provide two flow paths from the SIRW tank to the reactor coolant system.

i. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable. HCV-2914, 2934, 2974, and 2954 shall have power removed from their motor operators by locking open the circuit breakers in the power supply lines to the valve motor operators. FCV-326 shall be locked open.

(3) Whenever the reactor coolant system cold leg temperature is below 210°F and the reactor vessel head is installed, at least two (2) HPSI pump control switches shall be placed in pull-stop.

Whenever the reactor coolant system cold leg temperature is below 110°F and the reactor vessel head is installed, all three (3) HPSI pump control switches shall be placed in pull-stop.

In the event that no charging pumps are operable, a single HPSI pump may be taken from pull-stop and utilized for boric acid injection to the core.

Basis

The normal procedure for starting the reactor is to first heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing CEA's and diluting boron in the reactor coolant. With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable. During low power physics tests at low temperatures, there is a negligible amount of stored energy in the reactor coolant; therefore, an accident comparable in severity to the design basis accident is not possible and the engineered safeguards systems are not required.

The SIRW tank contains a minimum of 283,000 gallons of usable water containing at least 1700 ppm boron.(1) This is sufficient boron concentration to provide a shutdown margin of 5%, including allowances for uncertainties, with all control rods withdrawn and a new core at a temperature of 60°F.(2)

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 116.2 inch level corresponds to a volume of 825 ft^3 and the maximum 128.1 inch level corresponds to a volume of 895.5 ft^3.

Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

The operable status of the various systems and components is to be demonstrated by periodic tests. A large fraction of these tests will be performed while the reactor is operating in the power range.

(7) Direct communication between personnel in the control room and at the refueling machine shall be available whenever changes in core geometry are taking place.

(8) When irradiated fuel is being handled in the auxiliary building, the exhaust ventilation from the spent fuel pool area will be diverted through the charcoal filter.

(9) Prior to initial core loading and prior to refueling operations, a complete check out, including a load test, shall be conducted on fuel handling cranes that will be required during the refueling operation to handle spent fuel assemblies.

(10) A minimum of 23 feet of water above the top of the core shall be maintained whenever irradiated fuel is being handled.

If any of the above conditions are not met, all refueling operations shall cease immediately, work shall be initiated to satisfy the required conditions, and no operations that may change the reactivity of the core shall be made. However, refueling operations may commence and continue with less than 5 containment atmosphere and plant ventilation duct radiation monitors provided that gross, particulate and iodine monitors are monitoring the stack effluent. These three plant ventilation duct radiation monitors will initiate closure of the containment pressure relief, air sample and purge system valves and shall employ a one-out-of-three logic for the initiation of VIAS.

Irradiated fuel movement shall not be initiated before the reactor core has decayed for a minimum of 72 hours if the reactor has been operated at power levels in excess of 2% rated power.

Basis

The equipment and general procedures to be utilized during refueling operations are discussed in the FSAR. Detailed instructions, the above specifications, and the design of the fuel handling equipment incorporating built-in interlocks and safety features provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.(1) Whenever changes are not being made in core geometry one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The shutdown cooling pump is used to maintain a uniform boron concentration.

The shutdown margin as indicated will keep the core subcritical even if all CEA's were withdrawn from the core. During refueling operations, the reactor refueling cavity is filled with approximately 250,000 gallons of borated water. The boron concentration of this water (at least 1700 ppm boron) is sufficient to maintain the reactor subcritical by more than 5%, including allowance for uncertainties, in the cold condition with all rods withdrawn.(2) Periodic checks of refueling water boron concentration ensure the proper shutdown margin. Communication requirements allow the control room operator to inform the refueling machine operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

2.10.2 Reactivity Control Systems and Core Physics Parameters Limits

Applicability

Applies to operation of control element assemblies and monitoring of selected core parameters whenever the reactor is in cold or hot shutdown, hot standby, or power operation conditions.

Objective

To ensure (1) adequate shutdown margin following a reactor trip, (2) the MTC is within the limits of the safety analysis, and (3) control element assembly operation is within the limits of the setpoint and safety analysis.

Specification

(1) Shutdown Margin with T_{cold} >210°F

Whenever the reactor is in hot shutdown, hot standby or power operation conditions, the shutdown margin shall be $\geq$3.7% $\Delta k/k$. With the shutdown margin <3.7% $\Delta k/k$, initiate and continue boration until the required shutdown margin is achieved.

(2) Shutdown Margin with T_{cold} $\leq$210°F

Whenever the reactor is in cold shutdown conditions, the shutdown margin shall be $\geq$2.0% $\Delta k/k$. With the shutdown margin <2.0% $\Delta k/k$, initiate and continue boration until the required shutdown margin is achieved.

(3) Moderator Temperature Coefficient

The moderator temperature coefficient (MTC) shall be:

a. Less positive than $+0.2 \times 10^{-4}$ $\Delta\rho/°F$ including uncertainties for power levels at or above 80% of rated power.

b. Less positive than $+0.5 \times 10^{-4}$ $\Delta\rho/°F$ including uncertainties for power levels below 80% of rated power.

c. More positive than -2.3×10^{-4} $\Delta\rho/°F$ including uncertainties at rated power.

With the moderator temperature coefficient confirmed outside any one of the above limits, change reactivity control parameters to bring the extrapolated MTC value within the above limits within 3 hours or be in at least hot shutdown within 6 hours.

1. The total available shutdown margin may be reduced to 2% $\Delta k/k$ during the measurement of the shutdown CEA group reactivities, or

2. The total available shutdown margin may be reduced to the worth of the worst stuck CEA's during the measurement of the stuck CEA reactivity.

(ii) If the shutdown margin specified in part (i) above is not available immediately, initiate and continue boration until the requirements of 2.10.2(1) are met.

(iii) The shutdown margin specified in part (i) above shall be verified every 8 hour shift.

c. Moderator Temperature Coefficient

(i) The moderator temperature coefficient (MTC) requirements of 2.10.2(3) may be suspended during physics tests at less than 10^{-1}% of rated power.

(ii) If power exceeds 10^{-1}% of rated power, either:

1. Reduce power to less than 10^{-1}% of rated power within 15 minutes, or

2. Be in hot shutdown in 2 hours.

Basis

Shutdown Margin

A sufficient shutdown margin ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum shutdown margin of 3.7% $\Delta k/k$ is initially adequate to control the reactivity transient. Accordingly, the shutdown margin requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{cold} $\leq$210°F, the reactivity transients resulting from any postulated accident are minimal and a 2% $\Delta k/k$ shutdown margin provides adequate protection.

Control Element Assemblies

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum shutdown margin is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The specifications applicable to one or more CEA's that are determined to be untrippable or stuck, and to one or more misaligned CEA's that cannot be restored to within 12 inches of any other CEA in their group, require a prompt shutdown of the reactor since any of these conditions may be indicative of a possible loss of mechanical functional capability of the CEA system and in the event of an untrippable CEA, the loss of shutdown margin.

For small misalignments (<18 inches absolute) of the CEA's, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCO's and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCO's and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available shutdown margin, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the action statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in power. The one hour time is sufficient to 1) identify causes of a misaligned CEA, 2) take appropriate corrective action to realign the CEA's, and 3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment ($\geq$18 inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient

(b) If while operating under the provisions of part (a), the plant computer incore detector alarms become inoperable, operation may be continued without reducing power provided each of the following conditions is satisfied:

(i) A core power distribution was obtained utilizing incore detectors within 7 days prior to the incore detector alarm outage and the measured peak linear heat rate was no greater than 90% of the value allowed by (1) above.

(ii) The Axial Shape Index as measured by excore detectors remains within $\pm$.05 of the value obtained at the time of the last measured incore power distribution.

(iii) Power is not increased nor has it been increased since the time of the last incore power distribution.

(c) When the linear heat rate is continuously monitored by the excore detectors, withdraw the full length CEA's beyond the long term insertion limits of Specification 2.10.2.7. If the linear heat rate is exceeding its limits as determined by the Axial Shape Index, Y_I, being outside the limits of Figure 2-6, where 100 percent of the allowable power represents the maximum power allowed by the following expression:

$$\frac{L}{14.7} \times M \times N$$

where

1. L is the maximum allowable linear heat rate as determined from Figure 2-5 and is based on the core average burnup at the time of the latest incore power map.

2. M is the maximum allowable power level for the existing Reactor Coolant Pump combination.

3. N is the maximum allowable fraction of rated thermal power as determined by the F_{xy}^T limit curve of Figure 2-9 when monitoring by excore detectors. N = 1 when monitoring kw/ft using incore detectors.

(i) Restore the reactor power and Axial Shape Index, Y_I, to within the limits of Figure 2-6 within 2 hours, or

(ii) Be in at least hot standby within the next 6 hours.

2.0 LIMITING CONDITIONS FOR OPERATION
2.10 Reactor Core (Continued)

2.10.4 Power Distribution Limits

Applicability

Applies to power operation conditions.

Objective

To ensure that peak linear heat rates, DNB margins, and radial peaking factors are maintained within acceptable limits during power operation.

Specification

(1) Linear Heat Rate

The linear heat rate shall not exceed the limits shown on Figure 2-5 when the following factors are appropriately included:

1. Flux peaking augmentation factors are shown in Figure 2-8,

2. A measurement-calculational uncertainty factor of 1.058,

3. An engineering uncertainty factor of 1.03,

4. A linear heat rate uncertainty factor of 1.002 due to axial fuel densification and thermal expansion, and

5. A power measurement uncertainty factor of 1.02.

The measurement-calculational uncertainty in (1).2. above shall be increased, as necessary, pursuant to Specification 2.10.3(5)(a).

(a) When the linear heat rate is continuously monitored by the incore detectors, and the linear heat rate is exceeding its limits as indicated by four or more valid coincident incore detector alarms, either:

(i) Restore the linear heat rate to within its limits within one hour, or

(ii) Be in at least hot standby within the next 6 hours.

(2) Total Integrated Radial Peaking Factor

The calculated value of F_R^T defined by $F_R^T = F_R\ (1+T_q)$ shall be limited to ≤ 1.57. F_R is determined from a power distribution map with no part length CEA's inserted and with all full length CEA's at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. The azimuthal tilt, T_q, is the measured value of T_q at the time F_R is determined.

With $F_R^T > 1.57$ within 6 hours comply with any one of the following:

(a) Reduce power to bring power and F_R^T within the limits of Figure 2-9, withdraw the full length CEA's to or beyond the Long Term Steady State Insertion Limits of Specification 2.10.2(7), and fully withdraw the PLCEA's, or

(b) Reduce the allowed power of the axial power distribution DNBR monitoring curve of Figure 2-7 by the amount of $3.0 \times [(F_R^T/1.57)-1] \times 100$ percent of rated power, and reduce the values of 6563, 5122, and 4917 in the P_{var} equations of Specification 1.3(4) by the amount of $20 \times [(F_R^T/1.57)-1] \times 100$, withdraw the full length CEA's to or beyond the steady state insertions limits of Specification 2.10.2(7) and fully withdraw the PLCEA's, or

(c) Be in at least hot standby.

(3) Total Planar Radial Peaking Factor

The calculated value of F_{xy}^T defined as $F_{xy}^T = F_{xy}\ (1+T_q)$ shall be limited to ≤ 1.62. F_{xy} shall be determined from a power distribution map with no part length CEA inserted and with all full length CEA's at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects. The azimuthal tilt, T_q, is the measured value of T_q at the time F_{xy} is determined.

With $F_{xy}^T > 1.62$ within 6 hours comply with any one of the following:

(a) Reduce power to bring power and F_{xy}^T to within the limits of Figure 2-9, and withdraw the full length CEA's to or beyond the Long Term Steady State Insertion Limits of Specification 2.10.2(7) and fully withdraw the PLCEA's, or

NOTE: When power is reduced to comply with (a) above, and linear heat rate is being monitored using excore detectors, Figure 2-6 must be adjusted in accordance with Specification 2.10.4.(1).(c).

(b) Reduce the allowed power level of the axial power distribution linear heat rate monitoring limits of Figure 2-6 by the amount $2.22 \times [(F_{xy}^T/1.62)-1] \times 100$ percent of rated power, and reduce the positive and negative axial shape index trip limits of Figure 1-4 by the amount $.01 \times [(F_{xy}^T/1.62)-1] \times 100$ ASI units, withdraw the full length CEA's to or beyond the Long Term Steady State Insertion Limits of Specification 2.10.2(7) and fully withdraw the PLCEA's, or

(c) Be in at least hot standby.

(4) Azimuthal Power Tilt (T_q)

When operating above 70% of rated power, the azimuthal power tilt (T_q) shall not exceed 0.03.

(a) With the indicated azimuthal power tilt determined to be >0.03 but <0.10, correct the power tilt within two hours or determine within the next 6 hours and at least once per subsequent 8 hours, that the total integrated radial peaking factor, F_R^T, is within the limit of Specification 2.10.4(2) and that the total planar radial peaking factor, F_{xy}^T, is within the limit of 2.10.4(3), or reduce power to less than 70% of rated power within 8 hours of confirming T_q >0.03.

(b) With the indicated power tilt determined to be $\geq$.10, power operation may proceed up to 2 hours provided F_R^T and F_{xy}^T do not exceed the power limits of Figure 2-9, or be in at least hot standby within 6 hours. Subsequent operation for the purpose of measurement to identify the cause of the tilt is allowable provided:

(i) The power level is restricted to 20% of the maximum allowable thermal power level for the existing reactor coolant pump combination, and

(ii) The axial power distribution and thermal margin trip setpoints and the peripheral axial shape index limits of Figure 2-6 and 2-7 are adjusted in accordance with Specifications 2.10.4(2) and 2.10.4(3).

(5) DNBR Margin During Power Operation Above 15% of Rated Power

(a) The following DNB related parameters shall be maintained within the limits shown on Table 2-6.

(i) Cold Leg Temperature

(ii) Pressurizer Pressure

(iii) Reactor Coolant Flow

(iv) Axial Shape Index, Y_I

(b) With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce power to less than 15% of rated power within the next 8 hours.

Basis

Linear Heat Rate

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System, or the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the axial shape index with the operable quadrant symmetric excore neutron flux detectors and verifying that the axial shape index is maintained within the allowable limits of Figure 2-6 as adjusted by Specification 2.10.4.(1).(c) for the allowed linear heat rate of Figure 2-5, RC Pump configuration, and F_{xy}^T of Figure 2-9. In conjunction with the use of the excore monitoring system and in establishing the axial shape index limits, the following assumptions are made: (1) the CEA insertion limits of Specification 2.10.2.(6) and long term insertion limits of Specification 2.10.2(7) are satisfied, (2) the flux peaking augmentation factors are as shown in Figure 2-8, (3) the azimuthal power tilt restrictions of Specification 2.10.4.(4) are satisfied, and (4) the total planar radial peaking factor does not exceed the limits of Specification 2.10.4.(3).

The Incore Detector Monitoring System provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be continuously maintained within the allowable limits of Figure 2-5. The setpoints for these alarms include allowances, set in the conservative directions, for the factors listed in 2.10.4.(1).

The set points for the isolation function have been selected to limit radioactivity concentrations at the boundary of the restricted area to approximately 0.25 of 10 CFR 20 limits, assuming existence of annual average meteorology.

Each channel is supplied from a separate instrument a.c. bus and each auxiliary relay requires power to operate. On failure of a single a.c. supply, the A and B matrices will assume a one-out-of-two logic.

(4) Low Steam Generator Pressure

A signal is provided upon sensing a low pressure in a steam generator to close the main steam isolation valves in order to minimize the temperature reduction in the reactor coolant system with resultant loss of water level and possible addition of reactivity. The setting of 500 psia includes a ± 22 psi uncertainty and was the setting used in the safety analysis.[3]

(5) SIRW Tank Low Level

Level switches are provided on the SIRW tank to actuate the valves in the safety injection pump suction lines in such a manner so as to switch the water supply from the SIRW tank to the containment sump for a recirculation mode of operation after a period of approximately 24 minutes following a safety injection signal. The switchover point of 16 inches above tank bottom is set to prevent the pumps from running dry during the 10 seconds required to stroke the valves and to hold in reserve approximately 28,000 gallons of at least 1700 ppm borated water. The FSAR loss of coolant accident analysis[4] assumed the recirculation started when the minimum usable volume of 283,000 gallons had been pumped from the tank.

assumes a tripped condition (except high rate-of-change of power, high power level and high pressurizer pressure),[1] which results in a one-out-of-three channel logic. If in the 2 of 4 logic system of the reactor protective system one channel is bypassed and a second channel manually placed in a tripped condition, the resulting logic is 1 of 2. At rated power, the minimum operable high-power level channels is 3 in order to provide adequate power tilt detection. If only 2 channels are operable, the reactor power level is reduced to 70% rated power which protects the reactor from possibly exceeding design peaking factors due to undetected flux tilts and from exceeding dropped CEA peaking factors.

All engineered safety features are initiated by 2-out-of-4 logic matrices except containment high radiation which operates on a 1-out-of-5 basis.

The engineered safety features system provides a 2 of 4 logic on the signals used to actuate the equipment connected to each of the two emergency diesel generator units.

The rod block system automatically inhibits all CEA motion in the event a Limiting Condition for Operation (LCO) on CEA insertion, CEA deviation, CEA overlap or CEA sequencing is approached. The installation of the rod block system ensures that no single failure in the control element drive control system (other than a dropped CEA) can cause the CEA's to move such that the CEA insertion, deviation, sequencing or overlap limits are exceeded. Accordingly, with the rod block system installed, only the dropped CEA event is considered an AOO and factored into the derivation of the Limiting Safety System Settings and Limiting Conditions for Operation. With the rod block function out-of-service several additional CEA deviation events must be considered as AOO's. Analysis of these incidents indicates that the single CEA withdrawal incident is the most limiting of these events. An analysis of the at-power single CEA withdrawal incident was performed for Fort Calhoun for various initial Group 4 insertions, and it has been concluded that the Limiting Conditions for Operation (LCO) and Limiting Safety System Settings (LSSS) are valid for a Group 4 insertion of less than or equal to 15%.

References
(1) FSAR, Section 7.2.7.1

4.0 DESIGN FEATURES

4.4 Fuel Storage

4.4.1 New Fuel Storage

The new unirradiated fuel bundles will normally be stored in the dry new fuel storage rack with an effective multiplication factor of less than 0.9. The open grating floor below the rack and the covers above the racks, along with generous provision for drainage, precludes flooding of the new fuel storage rack.

New fuel may also be stored in shipping containers or in the spent fuel pool racks which have a maximum effective multiplication factor of 0.95 with Fort Calhoun Type C fuel and unborated water.

The new fuel storage racks are designed as a Class I structure.

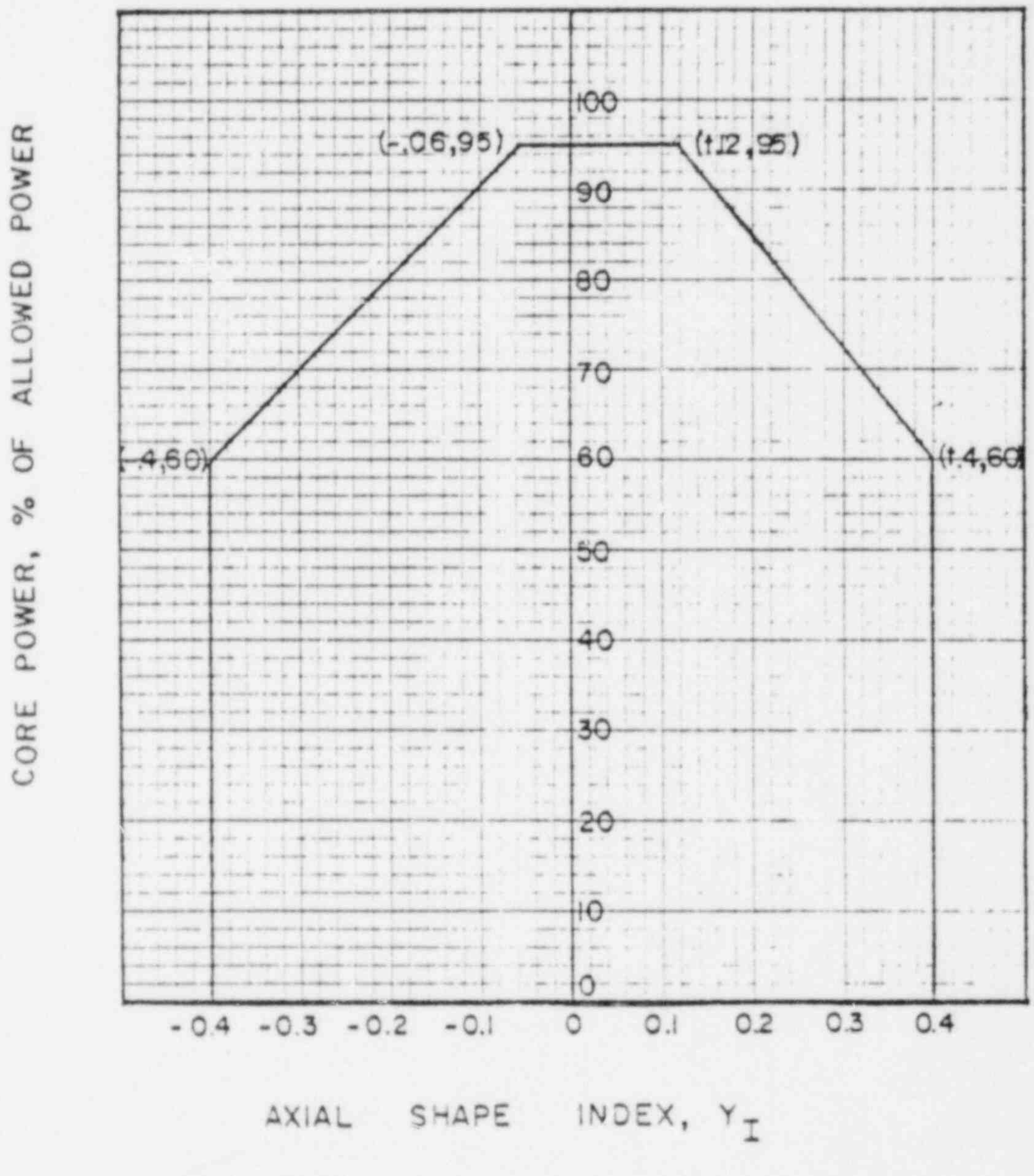
4.4.2 Spent Fuel Storage

Irradiated fuel bundles will be stored prior to off-site shipment in the stainless steel lined spent fuel pool. The spent fuel pool is normally filled with borated water with a concentration of at least 1700 ppm.

The spent fuel racks are designed to maintain fuel in a geometry which ensures an effective multiplication factor of 0.95 or less with new fuel containing not more than 40.1 grams of U-235 per axial centimeter of fuel assembly when the pool is filled with unborated water.

The spent fuel racks are designed as a Class I structure.

Normally the spent fuel pool cooling system will maintain the bulk water temperature of the pool below 120°F. Under other conditions of fuel discharge, the fuel pool water temperature is maintained below 140°F.



FORT CALHOUN TECHNICAL SPECIFICATIONS	LIMITING CONDITION FOR OPERATION FOR EXCORE MONITORING OF LINEAR HEAT RATES UP TO A MAXIMUM OF 14.7 KW/FT.	FIGURE 2-6

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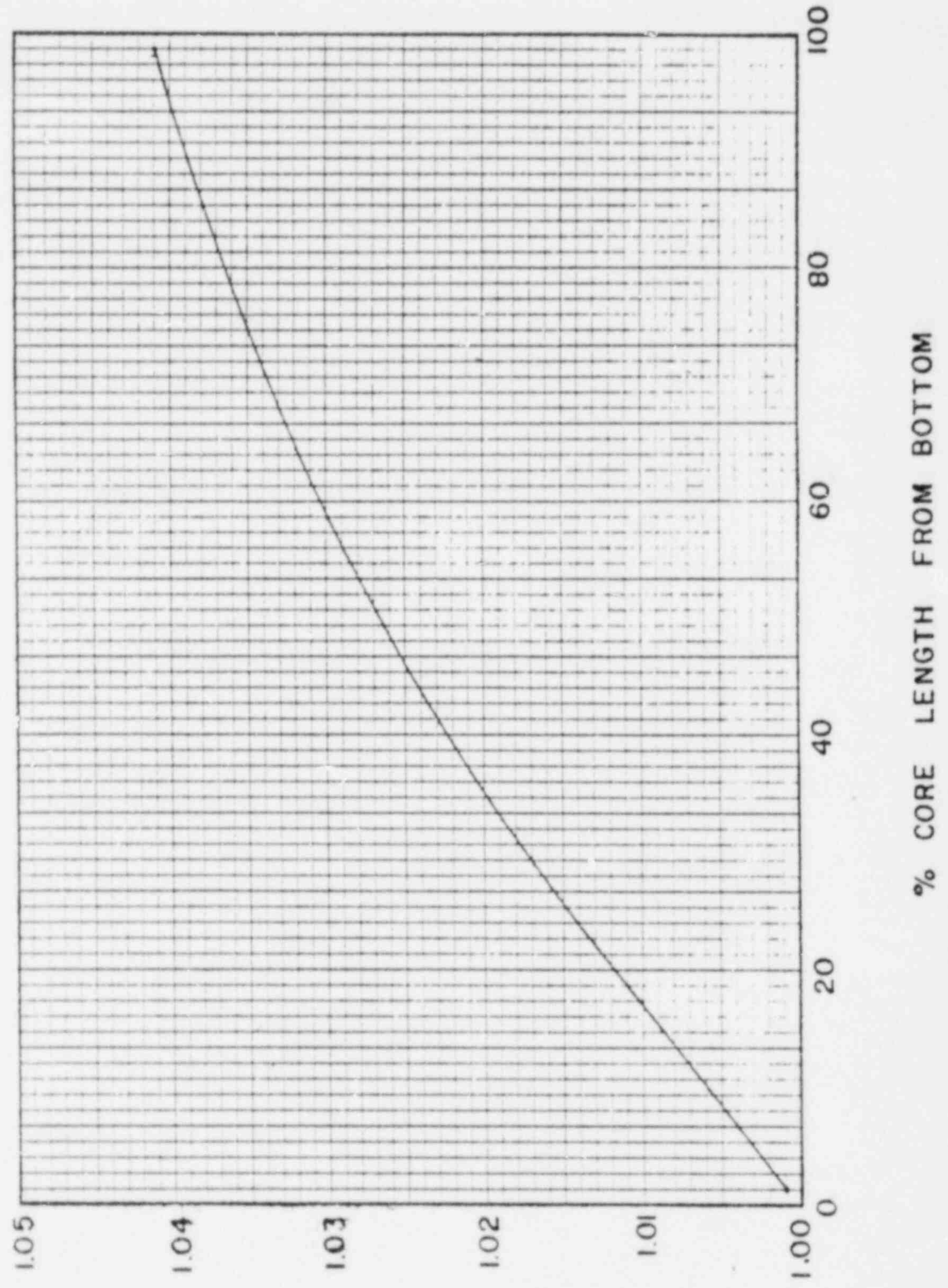

FORT CALHOUN TECHNICAL SPECIFICATIONS	FLUX PEAKING AUGMENTATION FACTORS	FIGURE 2-8

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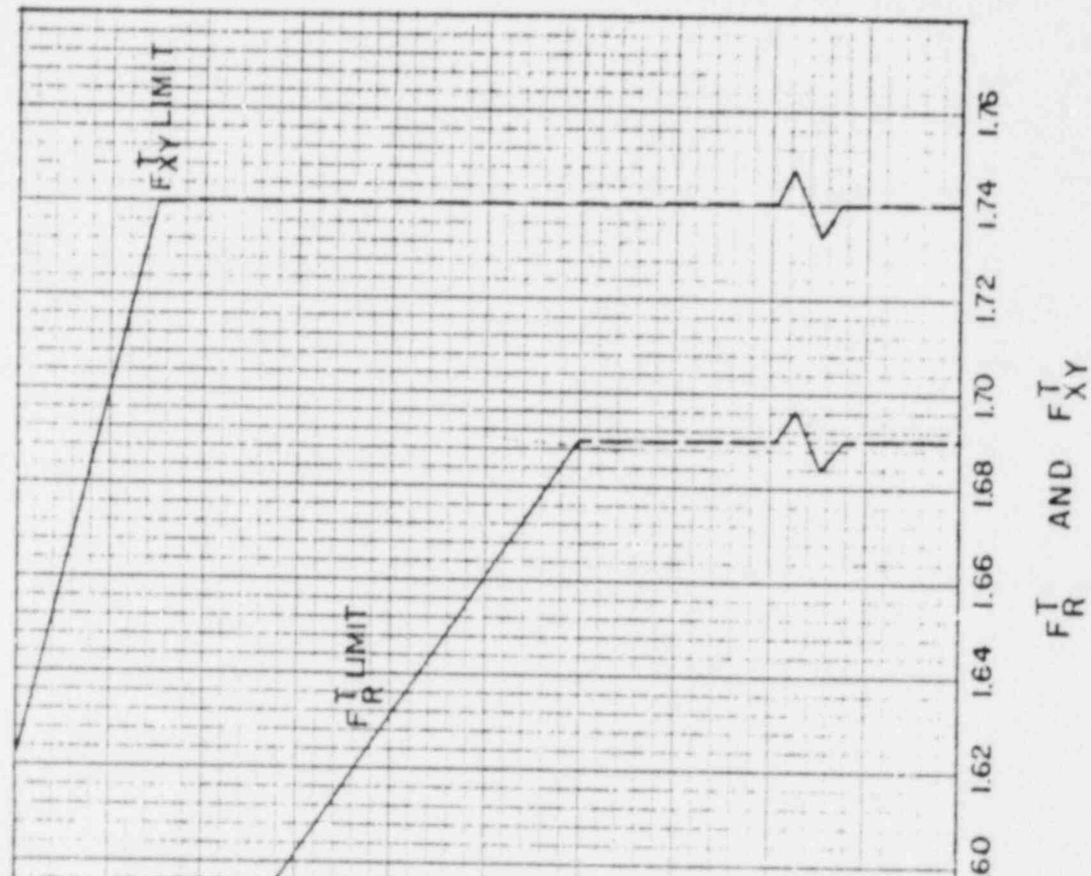

FORT CALHOUN TECHNICAL SPECIFICATIONS	F^T_R, F^T_{XY}, AND CORE POWER LIMITATIONS	FIGURE 2-9

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